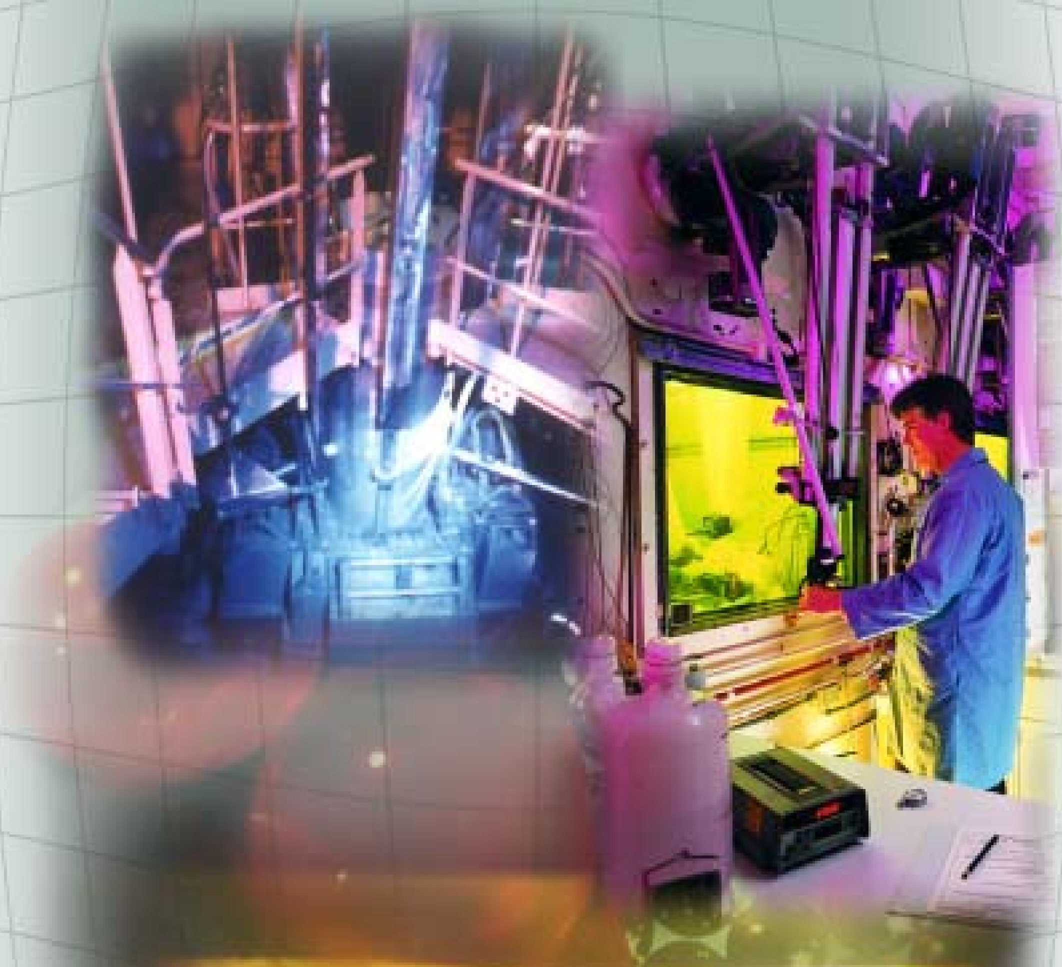
PNNL-SA-37780

International Nuclear Energy Research Initiative

Research Abstracts



April 2003

Office of Nuclear Energy, Science and Technology
U.S. Department of Energy

Background

In January 1997, the President of the United States requested his Committee of Advisors on Science and Technology (PCAST) to review the current national energy research and development (R&D) portfolio, and provide a strategy to ensure that the United States has a program to address the Nation's energy and environmental needs for the next century. In its November 1997 report responding to this request, the PCAST Energy R&D Panel determined that ensuring a viable nuclear energy option to help meet our future energy needs is important; and recommended that a properly focused R&D effort should be implemented by U.S. Department of Energy (DOE) to address the principal obstacles to achieving this option, including improving cost performance, increasing proliferation resistance, enhancing safety, and improving the waste management of nuclear energy systems.

In 1999, in response to the PCAST recommendations, DOE established the Nuclear Energy Research Initiative (NERI) to help overcome the principal technical and scientific issues affecting the future use of nuclear energy in the United States. Information on the NERI program including abstracts of the funded NERI projects are provided in the *Nuclear Energy Research Initiative 2002 Annual Report* and on the NERI website at http://neri.ne.doe.gov under the R&D Awards section.

Recognizing the importance of a focused program of international cooperation, PCAST issued a June 1999 report, entitled *Powerful Partnerships: The Federal Role in International Cooperation on Energy Innovation*, which highlights the need for an international component of the NERI program to promote "bilateral and multilateral research focused on advanced technologies for improving the cost, safety, waste management, and proliferation resistance of nuclear fission energy systems." The report further states that: "The costs of exploring new technological approaches that might deal effectively with the multiple challenges posed by conventional nuclear power are too great for the United States or any other single country to bear, so that a pooling of international resources is needed..."

The I-NERI component of NERI was established in FY 2001 in response to the PCAST recommendations. The I-NERI activity is enhancing the Department's ability to leverage its limited research funding with nuclear technology research funding from other countries while also providing the United States greater credibility and influence in international activities associated with the application of nuclear technologies.

To date, four I-NERI collaborative agreements have been established; the first between DOE and the Commisariat a l'Énergie Atomique (CEA) of France, the second between DOE and the Republic of Korea Ministry of Science and Technology (MOST), the third with the Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA) and the fourth with the European Commission. The primary U.S. client for the OECD/NEA program is the Nuclear Regulatory Commission (NRC), with DOE as a contributing partner. Since program inception, five projects with France, eleven with the Republic of Korea, and one with the Nuclear Energy Agency have been initiated. Discussions on collaboration are ongoing with Brazil, Canada, Japan, the Republic of South Africa, and the United Kingdom.

Goals & Objectives

In accomplishing its assigned mission, the following objectives have been established for the I-NERI program:

- Develop advanced concepts and scientific breakthroughs in nuclear energy and reactor technology to address and overcome the principal technical and scientific obstacles to the expanded use of nuclear energy worldwide.
- Promote collaboration with international agencies and research organizations to improve development of nuclear energy.
- Promote and maintain a nuclear science and engineering infrastructure to meet future technical challenges.

The Office of Nuclear Energy, Science, and Technology (NE) is coordinating a wide-ranging discussion among governments, industry, and the research community worldwide on the development of next-generation nuclear energy systems, known as the Generation IV Nuclear Energy Systems Initiative. I-NERI is a key collaboration mechanism for conducting R&D with international partners with a goal to develop Generation IV (Gen IV) nuclear energy systems.

Research Areas

The I-NERI Program sponsors innovative research and development in the following general areas:

- ♦ Next-generation (i.e., Generation IV) nuclear energy and fuel cycle technology.
- ♦ Next-generation nuclear plant designs with higher efficiency, lower cost, and improved safety and proliferation resistance.

- ♦ Innovative nuclear plant design, manufacturing, construction, operation, maintenance, and decommissioning technologies.
- ♦ Advanced nuclear fuels and materials.
- ♦ Fundamental nuclear science.

The specific workscope of each I-NERI collaboration is established by agreement between the U.S. Department of Energy (DOE) and the respective agency of the collaborating international country.

Program Control

Bilateral I-NERI agreements are normally established under existing or new "umbrella" agreements between the collaborating countries. The U.S. element of I-NERI is managed by NE who receives guidance from the Nuclear Energy Research Advisory Committee (NERAC). A counterpart agency of the collaborating country manages their participation.

A Bilateral I-NERI Steering Committee (BINERIC) made up of representatives from the United States and the collaborating country identifies specific research areas for mutually beneficial collaboration and decides other bilateral cooperation issues, such as required agreements, eligibility for participation, project selection processes, joint funding structure, and contractual vehicles. The BINERIC operates according to a set of guidelines approved by the collaborating countries. One executive agent (EA) from each country administers the I-NERI program under the guidance of the BINERIC.

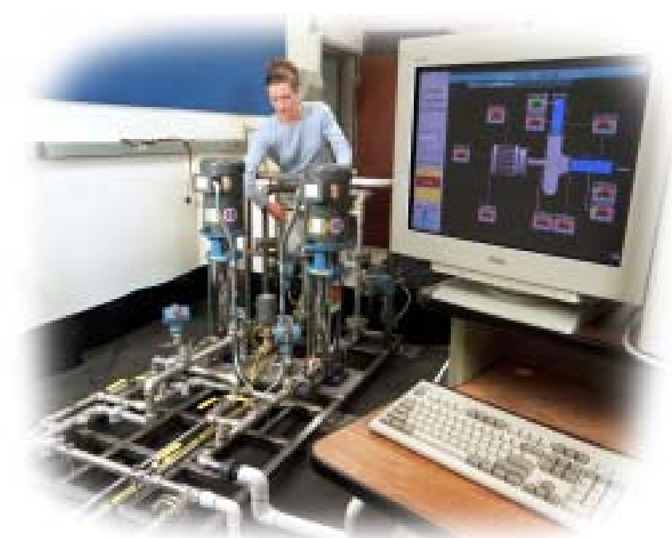
Project Selection

The I-NERI program incorporates competitive procurement processes for program activities. Competitive solicitations issued simultaneously by DOE and by the collaborating international agency result in submissions of collaborative researcher-initiated proposals. Eligibility includes laboratories, universities, and industry from the United States and collaborating countries. R&D organizations from these collaborating countries form research teams to develop integrated project proposals. Proposals are formally reviewed and the best potential collaborative projects are selected based on an integrated, peer review selection process. The specific solicitation workscope and the review and selection processes are tailored to the terms of each I-NERI agreement.

Peer review panels are selected in each country based on their technical expertise and capabilities in the fields for proposals that they will review. A common set of proposal evaluation criteria, which are established by the BINERIC, are used by each country in the peer review process. Separate peer reviews of the collaborative proposals are conducted by the United States and by the collaborating countries to determine the technical and scientific merit of each proposed project. NE receives a rank order list of the proposals from the peer review based on technical and scientific merit. A Federal programmatic review of the proposal is performed to ensure that proposed projects comply with Department policy and programmatic requirements. Analogous reviews are conducted simultaneously by the collaborating agency based on the rank order listing provided by their peer review panel. Final award selections of high merit, mutually beneficial proposals are made by the BINERIC in executive session via joint evaluation of the respective peer review results and recommendations.

Bilateral Collaborations

The I-NERI program has signed four bilateral agreements to date. The first agreement was signed in September 2000 with France's Commissariat a L'Energie Atomique (CEA). The second agreement was signed in May 2001 with the Republic of Korea's (ROK) Ministry of Science and Technology (MOST). A third bilateral collaboration involves Argonne National Laboratory and a consortium of ten international participants represented by the U.S. Nuclear Regulatory Commission (NRC) and the Organisation for the Economic Co-operation and Development (OECD) with offices throughout the world. The fourth agreement was signed in March 2003 with the European Commission.



United States/Republic of Korea Collaboration

The U.S./ROK collaboration is focused on advanced technologies for improving the cost, safety, and proliferation resistance of nuclear energy systems. U.S./ROK I-NERI projects have been competitively selected from researcher-initiated collaborative proposals based upon the results of an independent peer-evaluation process.

United States/France Collaboration

The U.S./CEA collaboration focuses on the development of Generation IV advanced nuclear system technologies that will enable the U.S. and France to move forward with leading-edge generic research that can benefit the range of reactor and fuel cycle designs anticipated in the future.

United States/OECD Collaboration

The U.S. has teamed with the OECD and a number of its members to do collaborative research to conduct reactor material experiments and associated analysis. The U.S. team members consist of NRC, DOE, and the Electric Power Research Institute. The OECD is an international organization that consists of 30 member countries. They assist member countries with harnessing technology change to boost economic growth and achieve other social objectives.

United States/European Commission Collaboration

The Department of Energy signed an agreement on March 6, 2003 with the European Atomic Energy Community (EURATOM) represented by the Commission of the European Communities. This collaboration focuses nuclear energy research including, but not limited to, innovated or revisited reactor concepts and plant optimization.

Planned Collaborations

Future collaborations are being considered with Canada, Brazil, Japan, Republic of South Africa, and the United Kingdom.

I-NERI Program Accomplishments

The I-NERI program effectively began in the second quarter of FY 2001, with an initial focus on development of international collaborations, program planning, and project procurements. Awards for the first set of I-NERI projects were made on the French collaboration at the end of FY 2001.

The primary programmatic accomplishments in FY 2001-2002, and planned accomplishments in FY 2003, are briefly described as follows:

FY 2001 Programmatic Accomplishments

- DOE signed collaborative I-NERI agreements with the Republic of Korea (May 2001) and France (July 2001).
- The U.S./France collaboration started with seven proposals resulting in the award of three projects in September 2001 and another in January 2002.
- The U.S./Republic of Korea program conducted a competitive procurement resulting in 21 proposals from which 6 projects were selected for FY 2002 awards.

FY 2002 Programmatic Accomplishments

- DOE and the Republic of Korea Minstry of Science and Technology (MOST) completed awards for 6 U.S./Republic of Korea projects involving 13 U.S. and 9 Republic of Korea participants from 15 universities, 4 national laboratories, and 4 industry partners.
- Added collaboration with the OECD/NEA under which one new project was awarded with funding provided by DOE, the NRC, and the Electric Power Research Institute (EPRI).
- Added one new project in the U.S./French collaboration on nuclear-based, thermo-chemical production of hydrogen, bringing total funded U.S./French projects to five.
- Conducted competitive procurement in the U.S./Republic of Korea collaboration resulting in 22 proposals from which five projects were selected for FY 2003 awards.

Planned FY 2003 Programmatic Accomplishments

- Complete FY 2002 annual project performance reviews for both U.S./France and U.S./Korean collaborations and confirm projects approved for ongoing support.
- Complete awards to the five proposals selected in the FY 2002 U.S. Korean competitive procurement.
- Initiate at least one new I-NERI collaboration with a new international partner.

I-NERI Program Budget (in millions)*			
FY 2001 Approp.	FY 2002 Approp.	FY 2003 Approp.	FY 2004 Request
$7.0	$8.5	$8.5	$9.0

*I-NERI budget is a portion of NERI appropriation

FY 2001 R&D Awards

The bilateral collaboration with France resulted in:

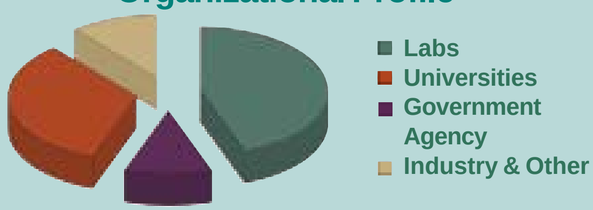
- 7 research proposals received by the Department and collaborating foreign organization from universities, national laboratories, and industry.
- 4 proposals were selected for award involving 8 U.S. and one foreign research organizations, all of which involved bilateral collaborations with multiple organizations.

- The duration of the awards is a three-year period and they will be funded annually. The total cost of these four projects is over $6 million.

- Award Organization Profile (four projects involving):
  - 4 National Laboratories
  - 3 Universities
  - 1 Foreign Government R&D Organization
  - 1 Industrial Organization

- INERI awards address the following R&D areas:
  - Generation IV Reactor Technology
  - Advanced Nuclear Fuels
  - Materials

U.S./France Collaboration Organizational Profile*



* based on number of involved organizations

FY 2002 R&D Awards

The bilateral collaboration with Korea resulted in:

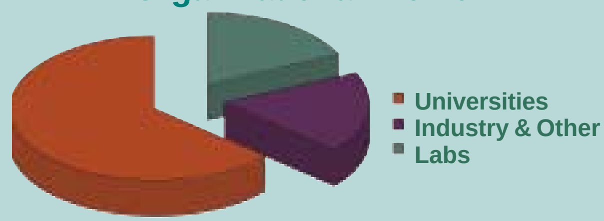
- 21 research proposals received by the Department and ROK's MOST from universities, national laboratories, and industry.

- 6 proposals were selected for award involving 13 U.S. and 9 foreign research organizations, all of which involved bilateral collaborations with multiple organizations.

- The duration of the awards is a three-year period and they will be funded annually. The total cost of these 6 projects is over $13 million.

- Award Organization Profile (6 projects involving):
  - 14 Universities
  - 4 Industrial Organizations
  - 4 National Laboratories

U.S./Korea Collaboration Organizational Profile*



* based on number of involved organizations

- FY 2002 INERI award areas:
  - Advanced Light-Water Reactor Technology
  - Advanced Instrumentation, Controls, and Diagnostics

In FY 2002 the U.S./French bilateral collaboration funded one new project.

- Objective is to test key processes for nuclear based thermo-chemical production of hydrogen.

- New project involves a U.S. industrial participant, a n ational lab, and a university.

The U.S./OECD collaboration funded one project in FY 2002.

- Scope of work encompasses experiments on molten core-concrete interactions in severe accidents.

- Work involves one U.S. national laboratory, EPRI, and several European collaborators.

Program Participants

Universities

University of Wisconsin
Purdue University
Ohio State University
Iowa State University
Pennsylvania State University
University of Manchester
University of Maryland
University of California-Santa Barbara
University of Michigan
Massachusetts Institute of Technology

National Laboratories

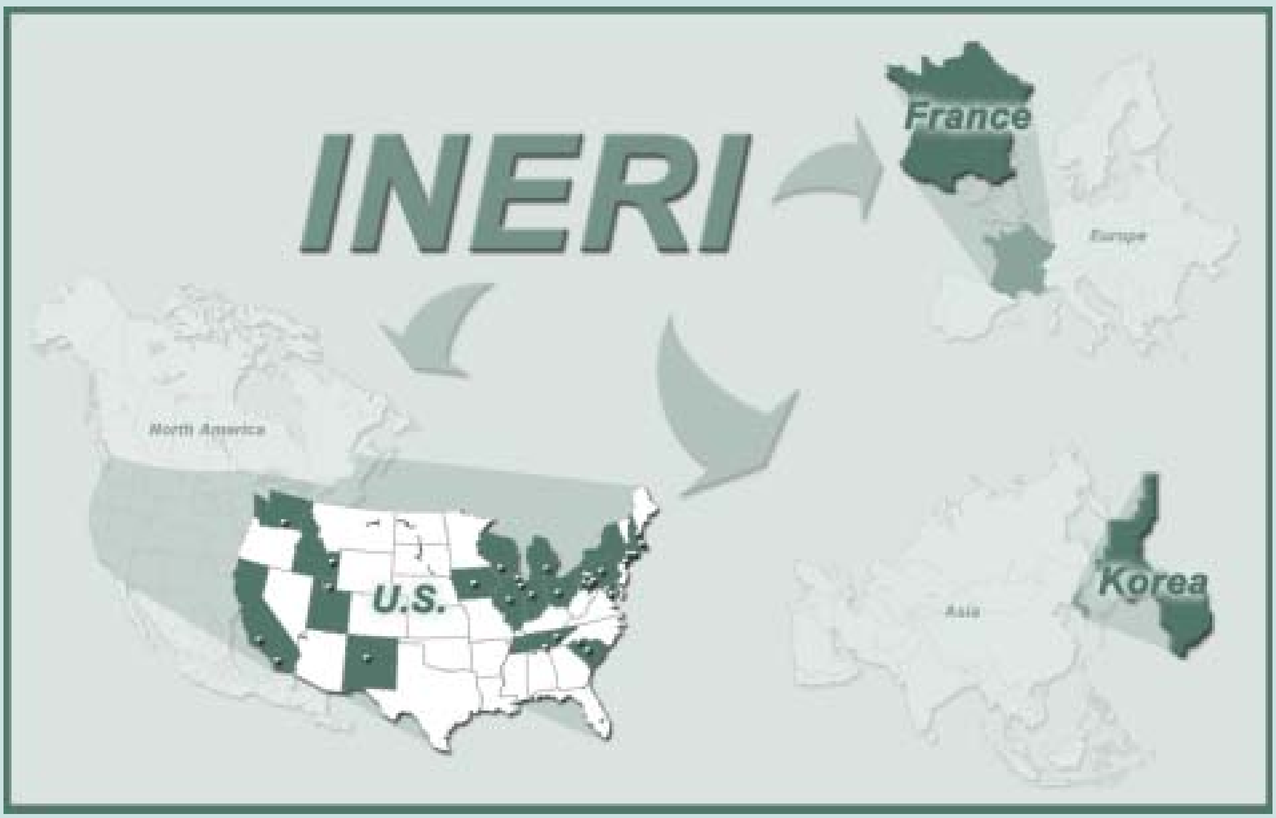
Argonne
Brookhaven
Oak Ridge
Idaho National Engineering and Environmental
Sandia

Industrial Organizations

Westinghouse Electric
General Atomics
Framatone-ANP, Lyon

International Collaborators

Korean Maritime University (Korea)
Chosun University (Korea)
Cheju University (Korea)
Seoul National University (Korea)
Pusan National University (Korea)
Chungnam National University (Korea)
Korea Atmoic Energy Research Institute (Korea)
Korean Electric Power Research Institute (Korea)
Korea Advanced Institute of Science & Technology (Korea)
Hanyang University
Organisation for Economic Cooperation and Development
Nuclear Energy Agency



U.S./France

Collaboration

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/CEA

ABSTRACTS

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U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/CEA

ABSTRACT

Development of Generation IV Advanced Gas-Cooled Reactors with Hardened/Fast Neutron Spectrum

Primary Investigator (U.S.): TYC Wei, Argonne National Laboratory

Primary Investigator (France): J. Rouault DEN/DER/SERI CEA Cadarache

Collaborators: Brookhaven National Laboratory; General Atomics; Massachusetts Institute of Technology; Oak Ridge National Laboratory; Framatome – ANP (Fra-ANP), Lyon

Project Number: 2001-002-F

Project Start Date: January 31, 2002

Project End Date: December 30, 2004

CEA and ANL have had similar reflections aimed at designing high temperature gas cooled fast/hardened spectrum reactors with a high degree of safety and an integrated fuel cycle. The work will be performed in a collaborative program to develop advanced gas-cooled reactors with a hardened or fast neutron spectrum which meet the Gen IV goals of economics, safety, sustainability and non-proliferation. To achieve this objective, the driving task for this project will be an advanced gas-cooled reactor concepts design/safety effort (WP1.2.1) with the associated integrated fuel cycle. A safety-in-the-design approach will be taken. There will be an exploratory phase where exploratory core/fuel forms/primary system concepts will be investigated to scope innovative design concepts. The initial focus will be on the Gen IV goal of safety with examination of various core, fuel form and primary system concepts which could achieve decay heat removal at low pressure conditions and Anticipated Transients without Scram (ATWS), essentially through passive or semi-passive cooling means. A certain level of active systems are not to be excluded, especially to control the initial phase of accidental transients. Sustainability concerns will also be factored into these designs, especially through the use of fast/hardened neutron spectrum and integrated fuel cycle. Economics and non-proliferation will be used as screening criteria. Those design concepts which are deemed of high potential will then be the subject of trade studies where secondary side, balance-of-plant, multifunction usage and size range issues will be considered. Simultaneously, a research and development plan will be drawn up to address the technical issues identified by the exploratory phase. Key among these will be the area of fuel development (WP1.2.2) and high temperature gas systems technology development (WP1.2.3). As the design trade studies are being conducted, parallel design support R&D work and safety support R&D work will be conducted to resolve these key technical issues. At the conclusion of the trade studies, design concepts which are selected will become the subject of a point design characterization effort where key design parameters will be established. The results of the project effort can therefore be utilized in a conceptual design study. To achieve this end goal, it has been mutually decided between CEA and ANL that the optimum R&D route would be to closely integrate the three CEA/ANL work packages into one comprehensive work plan. The three packages are:

(1) WP1.2.1: Advanced Gas-Cooled Reactor Concepts
(2) WP1.2.2: Fuel Development and Associated Fuel Cycle Processes (Treatment and Refabrication)
(3) WP1.2.3: High-Temperature Gas Systems Technology (Postponed for future consideration)

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/CEA

ABSTRACT

Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels

Primary Investigator (U.S.): David Petti, Idaho National Engineering and Environmental Laboratory

Primary Investigator (France): Philippe Martin, DEN/DEC/SESC CEA

Collaborators: Massachusetts Institute of Technology

Project Number: 2001-003-F

Project Start Date: September 29, 2001

Project End Date: September 30, 2004

High-temperature gas-reactor technology is achieving a renaissance around the world. Preliminary research has concluded that this technology has an excellent opportunity to satisfy the safety, economic, proliferation, and waste disposal concerns that face nuclear electric generating technologies. The potential economics of gas reactors are attractive enough that development continues in a number of countries. Small gas research reactors have been built in Germany, Japan and China. Russia and the United States have a project to develop a Modular High Temperature Gas Reactor (prismatic type) to burn excess plutonium. An ambitious project in this area is being pursued by a large utility in South Africa (ESKOM), which is proposing to build a 110 MWe pebble-bed gas reactor for commercial electric generation within the next five years. Fast reactors with gas coolant and a self-sufficient fuel cycle may also be needed in the future.

Many technical issues must, however, be addressed before construction of gas-cooled reactors becomes commercially viable. One of the most important issues is the behavior of coated particle gas reactor fuels during normal, off-normal, and accident conditions. The experience with coated-particle gas reactor fuels over the last 30 years is that some have performed well while others have performed poorly. No consensus has been reached in explaining the observed differences in fuel behavior.

The classical gas reactor coated-particle fuel is a spherical layered composite of microscopic dimensions. It has a fissile fuel kernel, generally made of UO_2 or UC_2, or UCO, that is surrounded by a porous graphite buffer layer that absorbs radiation damage, allows space for fission gases produced during irradiation, and resists kernel migration at high temperature. Surrounding the buffer layer is a layer of dense pyrolytic carbon, an SiC layer, and one or two dense outer pyrolytic carbon layers. The pyrolytic carbon layers act to protect the SiC layer, which is the primary pressure boundary for the micro-sphere. The inner pyrolytic carbon layer also protects the kernel from corrosive gases that are present during the deposition of the SiC layer. This layer arrangement is known as the TRISO coating system. Each micro-sphere acts as a mini pressure vessel, a feature that is intended to impart robustness to the gas reactor fuel system.

Compared to light water reactor and liquid metal reactor fuel forms, the behavior of coated-particle fuel is inherently more multi-dimensional. This is due to such anomalies as potential shrinkage cracking in the Inner Pyrocarbon (IPyC) layer, potential de-bonding between coating layers, and some degree of particle asphericity. Fuel behavior is

2001-003-F (continued)

further complicated by statistical variations in fuel physical dimensions and material properties from particle to particle due to the nature of the fabrication process. Previous models of coated fuel have typically been simplified one-dimensional models designed to perform simulations on large populations of particles at a reasonable speed.

The challenge addressed in this project is to produce an integrated mechanistic model for coated-particle fuel that accurately accounts for the multi-dimensional behavior of the coating layers, chemical changes of the fuel kernel during irradiation, and statistical variations in the dimensions and properties of the coating layers. The advent of powerful personal computers and the advancements in fundamental modeling of materials science processes should make this possible. Results from the model will facilitate the design of coated particle fuel for the gas reactor or other particle fuel applications.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/CEA

ABSTRACT

OSMOSE – An Experimental Program for Improving Neutronic Predictions of Advanced Nuclear Fuels

Primary Investigator (U.S.): Raymond Klann,
Argonne National Laboratory

Primary Investigator (France): Jean-Pascal Hudelot,
DRN/DER/SPEx/LPE CEA

Collaborators: University of Michigan

Project Number: 2001-006-F

Project Start Date: September 29, 2001

Project End Date: September 30, 2004

The objective of this collaborative program with the French CEA is to measure very accurate integral reaction rates in representative spectra for the actinides important to future nuclear system designs and to provide the experimental data for improving the basic nuclear data files. These data will support advanced reactors designed for transmutation of waste or Pu burning, sub-critical systems such as found in advanced accelerator applications, and the waste disposal and treatment program in the area of criticality safety. This program is very generic, in the sense that it will measure these reaction rates over a broad range of isotopes and spectra and will be used to provide guidance to all nuclear data programs in the world. These data will provide information valuable to a large number of projects as noted above.

The design of nuclear systems has shifted over the years from a "test and build" approach to a much more analytical methodology based on the many advances in computational techniques and nuclear data. To a large extent current reactors can be calculated almost as well as they can be measured. This is due in particular to the high quality nuclear data available for the few major isotopes which dominate the neutronics of these systems. Nevertheless, most future nuclear systems concepts and advanced fuels development programs currently under way use significant quantities of minor actinides to address modern day issues such as proliferation resistance and low cost. For example, proliferation resistant reactors and fuels are typically based on ^{232}Th and ^{233}U. High burnup fuels contain large quantities of americium and curium. Systems designed for plutonium and minor actinide burning are very sensitive to uncertainties in Americium and Curium data. There are also several other programs where the minor actinide data are essential. These include the Accelerator Transmutation of Waste concepts, and Burnup Credit programs.

The need for better nuclear data has been stressed by various organizations throughout the world, and results of studies have been published which demonstrate that current data are inadequate for designing the projects under consideration. In particular, a Working Party of the OECD has been concerned with identifying these needs and has produced a detailed High Priority Request List for Nuclear Data. The first step in obtaining better nuclear data consists of measuring accurate integral data and comparing it to integrated energy dependent data: this comparison provides a direct assessment of the effect of deficiencies in the differential data. Several US and international programs have indicated a strong desire to obtain accurate integral reaction rate data for improving the major and minor actinide cross sections. Specifically, these include: ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{242}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, and ^{247}Cm. Data on the major actinides

2001-006-F (continued)

(i.e. ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am) are reasonably well-known and available in the Evaluated Nuclear Data Files - (JEF, JENDL, ENDF-B). However, information on the minor actinides (i.e.^{232}Th, ^{233}U, ^{237}Np, ^{238}Pu, ^{242}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, and ^{247}Cm) is less well-known and considered to be relatively poor in some cases, having to rely on models and extrapolation of few data points. This is mainly due to the difficulty of obtaining relatively pure samples of sufficient quantity (up to about one gram) to perform reliable reaction rate measurements.

The French Atomic Energy Commission (CEA) has also recognized the need for better data and has launched an ambitious program aimed at measuring the integral absorption rate parameters in an experimental facility located at the Cadarache Research Center. A complete analytical program is associated with the experimental program and aims at understanding and resolving potential discrepancies between calculated and measured values. The final objective of the program is to reduce the uncertainties in predictive capabilities to a level acceptable to core designers and government regulators.

Argonne National Laboratory has expertise in these areas. In the past, ANL teams have developed very accurate experimental techniques and will strongly enhance the content of the experimental program. Furthermore, current ANL staff have heavily participated in the development of the French experimental and analytical program, and have contributed to the computational tools used by the French teams.

In this program, ANL staff will participate in the experimental measurements made in the MINERVE reactor at Cadarache, and all the data will be available to the U.S. in exchange for this participation.

ANL also has the facilities to perform independent measurements using the NRAD reactor. The characteristics of the NRAD reactor are significantly different from the MINERVE reactor and the results will complement the French measurements. NRAD also offers the flexibility to tailor the neutron spectrum, thereby, broadening the useful energy range of the experiments. In addition to the samples prepared by the French (and loaned to ANL for measurements), ANL will be preparing additional samples for measurements, depending on availability.

This three-year project has three critical outcomes:

1. High quality experimental data representative of the major and minor actinides will be made available to the US programs.

2. The US neutronics and criticality safety codes will be validated for reactivity effects from the major and minor actinides.

3. Potential deficiencies in US nuclear data and analysis tools will be identified.

In addition to the critical outcomes from the project, there are general benefits to the nuclear program in the United States. Specifically, by cross-calibration with measurements in the NRAD facility, the uncertainties in the data will be better known and deficiencies in the cross-section data will become apparent. This will lead to a better understanding of the available cross-section data and which areas need further development and research plans. This project also intends on involving a graduate student in the measurement and analysis tasks. By introducing young experimentalists to the project through key involvement in tasks, expertise is developed within the United States. This is vital since there are very few remaining experimentalists in this area.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/CEA

ABSTRACT

Nano-Composited Steels for Nuclear Applications

Primary Investigator (U.S.): Roger Stoller, Oak Ridge National Laboratory

Primary Investigator (France): A. Alamo, CEA Saclay

Collaborators: University of California, Santa Barbara

Project Number: 2001-007-F

Project Start Date: September 29, 2001

Project End Date: September 30, 2004

The successful development of advanced nuclear technologies embodied in the overall concept of Generation IV reactors will provide major challenges to the field of materials science. An underlying theme common to all systems under consideration is the critical importance of developing fuel cladding and structural materials that outperform the best of the materials currently code-qualified for in-core applications. For example, the development of alloys with greatly improved creep strength, fracture resistance, and oxidation resistance combined with a high degree of tolerance for radiation damage would enable systems to be designed with higher operating temperatures and extended component lifetimes.

Alloy development efforts in Japan, Europe, and the U.S. have shown that nano-composited versions of the $Fe^{-8}Cr$ ferritic-martensitic and $Fe^{-13}Cr$ ferritic steels have the potential for developing into a new class of creep and oxidation-resistant steels with outstanding swelling resistance that could be deployed in Generation IV reactor systems. Work is continuing at various institutions to address issues such as anisotropic properties, thermal embrittlement, strength loss associated with microstructural instabilities, and compatibility with liquid and gaseous environments.

This collaborative proposal seeks to develop a scientific knowledge base on the fundamental deformation and fracture characteristics of several carefully selected representative nanocomposited steels strengthened by fine scale oxide particles or atom clusters rich in yttria, titanium, and oxygen. This information will be combined with a study of the effects of neutron irradiation on microstructural stability and mechanical behavior to provide a solid foundation for further development of these innovative materials for nuclear applications.

Miniaturized tensile, creep, and fracture testing techniques will be used, coupled with microstructural characterization to define a limited set of promising composition/microstructures for neutron irradiation studies. Nano-scale structural characterization will be carried out to provide input to microstructure-property modeling activities. Extensive use of Ashby deformation and fracture mapping will be made to characterize the dominant regimes of deformation and fracture processes. A preliminary assessment will be made of the radiation response of a limited number of promising composition/microstructures; irradiations will be carried out using experimental facilities at Phènix, HFIR, or JOYO. The impact of neutron irradiation on the deformation and fracture regimes will be mapped. Relationships between the radiation-modified microstructure and the dominant modes of deformation and fracture will be modeled. This information will be integrated with that obtained from other programs to develop a thorough evaluation of the potential of these materials for expanding the operating temperature limits of fission reactor systems.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/CEA

ABSTRACT

High Efficiency Hydrogen Production From Nuclear Energy: Laboratory Demonstration of S-I Water-Splitting

Primary Investigator (U.S.): Paul S. Pickard, Sandia National Laboratory

Primary Investigator (France): Stephen Goldstein, CEA Saclay

Project Number: 2002-001-F

Project Start Date: September 1, 2002

Project End Date: October 31, 2005

Application of nuclear energy to the efficient and economic production of hydrogen has the potential to revolutionize our use of transportation fuels and provide a path to a more secure energy future. Nuclear hydrogen production would provide an essentially carbon emissions free source of transportation fuels, dramatically reduce dependence on fossil fuels, and open a new area of application for nuclear energy that may eventually exceed the use of nuclear power for electricity. One of the most promising approaches to achieve this goal is the development of high efficiency thermochemical cycles driven by high temperature advanced reactors. These cycles potentially offer the most energy efficient and economically scalable path to the levels of production needed for a future hydrogen economy.

The most promising thermochemical cycle for nuclear application identified in recent studies is the Sulfur-Iodine cycle. The Sulfur-Iodine cycle involves three component chemical reaction systems to achieve thermochemical water-splitting cycle for production of hydrogen. These systems respectively create H_2SO_4 and HI and separate the acids, carry out the reactive distillation decomposition of HI and the concentration/decomposition of H_2SO_4. This current NERI project reviewed the substantial world data base of thermochemical water-splitting reactions, and selected the Sulfur-Iodine cycle as the most promising from an initial compilation of 115 thermochemical hydrogen cycles. This research will provide a detailed flow sheet design of the nuclear - matched Sulfur-Iodine water-splitting cycle and a model of process chemistry and heat and material balances.

The Department of Physico-Chemistry in the Nuclear Energy Directorate of the French Commissariat à l'Énergie Atomique (CEA) has also done a preliminary evaluation of different methods to produce hydrogen from nuclear energy. They have identified the Sulfur-Iodine thermochemical water-splitting process as one of the leading candidates.

The next step in achieving the nuclear hydrogen goal is laboratory demonstration of the Sulfur – Iodine cycle, which is the focus of this INERI proposal. Based on these preliminary evaluations, the French CEA and US DOE teams will collaborate on this next step. The objective is to demonstrate the key technology elements of the Sulfur – Iodine system with sufficient fidelity to allow an engineering and economic assessment of the viability of this approach for nuclear hydrogen production.

2002-001-F (continued)

These next steps involve demonstration of the three chemical reaction systems of the S-I cycle. These component reactions are the Prime, or Bunsen, Reaction, the HI Decomposer and the H_2SO_4 Concentrator/Boiler/Decomposer. We propose to design, build and test laboratory-scale demonstrations of these systems. This will demonstrate efficient operation of the Bunsen Reaction that is fundamental to high cycle efficiency, demonstrate successful operation of the HI reactive distillation column that is key to a cost-effective S-I cycle, and will demonstrate the heat exchanger materials technology in the corrosive H_2SO_4 environment that is necessary for high process availability.

The tasks needed to demonstrate the Sulfur- Iodine cycle will be performed in a fully coordinated approach that utilizes the technical capabilities of the CEA and U.S. participants. Sandia National Laboratories will have responsibility for coordination of the effort and will build and test the front-end system that boils and concentrates sulfuric acid to form sulfur dioxide vapor for reaction with iodine. The French team will design and build the prime (Bunsen) reactor and the General Atomics team will design and build the system that decomposes hydroiodic acid vapor (HI) to produce hydrogen. The design of the component reaction sections will be based on an integrated flow sheet analysis that will be developed for a fully integrated, closed loop demonstration system.

The three laboratory-scale units will be sized and designed so that they may subsequently be connected together with other S-I cycle components and integrated into a complete system for integrated, closed-loop testing. Integration of the component reaction sections, and testing and operation of the integrated closed loop demonstration experiments would be the next phase of Sulfur – Iodine demonstration. This closed loop demonstration would require additional resources to complete. A detailed option for accomplishing this closed loop integration step is being developed collaboratively by the US and French partners, and will be available by August 7, 2002. The closed loop demonstration is an essential step in preparation for a full-scale pilot plant demonstration in the future.

U.S./Republic of Korea

Collaboration

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/ROK

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DOE/ROK

ABSTRACT

Fundamentals of Melt-Water Interfacial Transport Phenomena: Improved Understanding for Innovative Safety Technologies in ALWRs

Primary Investigator (U.S.): Michael Corradini, University of Wisconsin

Primary Investigator (Republic of Korea): KH Bang, Republic of Korea Maritime University

Collaborators: Argonne National Laboratory

Project Number: 2002-008-K

Project Start Date: January 15, 2002

Project End Date: December 30, 2004

The interaction and mixing of high-temperature melt and water is the important technical issue in the safety assessment of water-cooled reactors to achieve ultimate core coolability (conventional as well as advanced LWRs). For specific advanced light water reactor (ALWR) designs, deliberate mixing of the core-melt and water is being considered as a mitigative measure, to assure core coolability. The development and evaluation of such an innovative safety concept are hampered by the lack of fundamental understanding of the interfacial transport phenomena involved in melt-water mixing during the transient quenching process. The goal of this work is then to enhance the fundamental understanding of melt-water interfacial transport phenomena, thus enabling the development of innovative safety technologies for advanced LWRs. To achieve the goal of improved understanding of melt-water mixing and associated interfacial transport phenomena, the objectives of this research will be:

1) Adapt the existing experimental facilities at the University of Wisconsin-Madison (UW) and Argonne National Laboratory (ANL) for the characterization of transient cool down of melt mixed with water.

2) Measure the cool down behavior of the melt-water mixing zone by thermal mapping of this multiphase, multicomponent system (ANL).

3) Measure the flow regime and interfacial area behavior of the melt-water multiphase, multi-component mixture by the use of innovative real-time X-ray imaging (UW).

4) Develop and integrate analytical models of interfacial transport phenomena in the framework of a computer model, including separate-effects experimental studies (Korean researchers at the Korea Maritime University).

5) Demonstrate the applicability of fundamental knowledge to the development of a novel safety concept of ex-vessel core debris coolability by (all participants): developing the scaling logic for this large scale application, analyzing past and current experiments that involve quenching of a simulant oxidic melt by water injection, assess needs for larger scale melt-water mixing tests in future R&D work.

The intent of these research objectives is to provide better understanding of the process of melt-water mixing and interactions specifically in representative reactor geometries and with simulant materials to demonstrate the capability to provide core coolability to achieve a safe stable state.

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DOE/ROK

ABSTRACT

The Numerical Nuclear Reactor for High-Fidelity Integrated Simulation of Neutronic, Thermal-Hydraulic, and Thermo-Mechanical Phenomena

Primary Investigator (U.S.): David Weber, Argonne National Laboratory

Primary Investigator (Republic of Korea): HG Joo, Republic of Korea Atomic Energy Research Institute (KAERI)

Collaborators: Purdue University; Seoul National University

Project Number: 2002-010-K

Project Start Date: December 11, 2001

Project End Date: December 30, 2004

This ANL- and KAERI-led collaborative project will develop a comprehensive high-fidelity reactor-core modeling capability that could be used for detailed analysis of both current and advanced reactor designs. The work will involve the coupling of advanced numerical models such as computational fluid dynamics (CFD) for thermal-hydraulic calculations, whole-core discrete integral transport for neutronics calculations, and thermo-mechanical techniques for structural calculations. To achieve highest fidelity possible, the product code will be designed to run on massively parallel high performance computers, although it will be structured to run efficiently on a cluster of workstations or PCs and scalable systems.

The goals of this project are to:

- Assess the various methods and code systems that could provide the detailed information pertinent to the project.
- Demonstrate and evaluate high-fidelity CFD, neutronic, and thermo-mechanical techniques for whole-core transient applications.
- Develop efficient numerical methodology for coupling the neutronic, thermal-hydraulic, and thermo-mechanical capabilities that are based on high-fidelity methods.
- Evaluate performance of capability on different computer platforms, from small cluster of workstations to massively parallel machines.
- Validate analysis capability against selected experiments and numerical test cases.

The advanced computer simulation capabilities developed under this project would provide a verifiable computational tool for calculation of detailed numerical estimates of the neutronic, thermal-hydraulic, thermo-mechanical performance of current and advanced nuclear systems. The coupled code system would also enable analysts to perform intensive studies on the operational and safety characteristics of various design alternatives, and to compare results obtained with presently available tools to those from this high-fidelity capability.

2001-010-K (continued)

The project team includes international experts from the U.S. and ROK in the area of coupled code development and applications. The U.S. effort is led by ANL, while the ROK team is lead by KAERI. Two universities, Purdue (U.S.) and Chosun (ROK), will also participate in the project.

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DOE/ROK

ABSTRACT

Advanced Computational Thermal Fluid Physics (CTFP) and its Assessment for Light Water Reactors and Supercritical Reactors

Primary Investigator (U.S.): Don McEligot, Idaho National Engineering and Environmental Laboratory (INEEL)

Primary Investigator (Republic of Korea): JY Yoo, Seoul National University

Collaborators: Iowa State University; Pennsylvania State University; University of Maryland; University of Manchester; KAIST

Project Number: 2002-016-K

Project Start Date: December 11, 2001

Project End Date: December 30, 2004

The objective of this proposed Korean/US collaboration of coupled computational and experimental studies addresses fundamental science and engineering to develop supporting knowledge required for reliable approaches to new and advanced light water and supercritical reactor (ALWR and SCR) designs for improved performance, efficiency, reliability, enhanced safety and reduced costs and waste. It will provide basic thermal fluid science knowledge to develop increased understanding for the behavior of superheated and supercritical systems at high temperatures, application and improvement of modern computation and modeling methods and incorporation of enhanced safety features. The project promotes, maintains and extends the nuclear science and engineering infrastructure to meet future technical challenges in design and operation of high efficiency reactors and nuclear plant safety; it brings recognized thermal fluid mechanics authorities, Profs. Lee, Park, Pletcher, Yoo and Wallace and their students, into the nuclear science and engineering research community.

This basic thermal fluids research applies first principles approaches (Direct Numerical Simulation - DNS and Large Eddy Simulation - LES) coupled with experimentation (heat transfer and fluid mechanics measurements). Turbulence is one of the most important unresolved problems in engineering and science, particularly for the complex geometries and property variation occurring in these advanced reactor systems and their passive safety systems. DNS, LES and differential second-moment closures (DSM or Reynolds-stress models) are advanced computational concepts in turbulence "modeling" whose development will be extended to treat complex geometries and severe property variation for design and safety analyses of advanced light water and supercritical reactors.

Prof. Pletcher will extend LES to generic idealizations of such geometries; Prof. Yoo will support these studies with DNS. Prof. Wallace will develop miniaturized multi-sensor probes to measure turbulence components in supercritical flows. INEEL will obtain fundamental turbulence and velocity data for generic idealizations of the complex geometries of these advanced reactor systems and, with Profs. Yoo and Lee, develop experiments on the effects of property variation in superheated and supercritical flows. Prof. Park will develop DSM models and evaluate the suitability of other proposed RANS (Reynolds-averaged Navier-Stokes) models by application of the DNS, LES and experimental results. Profs. Hochreiter, formerly of Westinghouse, and Jackson will provide industrial insight and thermal-hydraulic data needs and will review the results of the proposed studies for application to realistic designs and their predictive safety and design codes.

2001-016-K (continued)

The proposed study will extend current NERI efforts for HTGRs to light water and supercritical reactors; it will not duplicate research underway on the current project. The unique INEEL Matched-Index-of-Refraction flow system, the World's largest such facility, will be applied to obtain fundamental data on flows through complex geometries important in the design and safety analyses for advanced light water and supercritical reactors. Successful completion of the proposed work will provide the following new fundamental and engineering knowledge, which is not presently available in the literature:

- Measurements of wall friction and loss coefficients in supercritical flows through non-circular ducts
- Time-resolved basic measurements of turbulence quantities (e.g., turbulence kinetic energy and Reynolds stresses) in superheated and supercritical flows with large property variation.
- Time-resolved data plus visualization of flow around obstructions, such as grid spacers in closely-packed reactor cores
- Separation of effects of phenomena — buoyancy, property variation and acceleration – occurring in strongly-heated superheated and supercritical flow to evaluate their individual importances and consequent flow behavior (by application of LES and DNS since this separation is not feasible in experiments with superheated and supercritical fluids at appropriate conditions).
- Application of DNS and LES for the first time to complex turbulent flows occurring in advanced light water and supercritical reactors
- Fundamental data of internal turbulence distributions for assessment and guidance of CTFD codes proposed for design and safety analyses for these advanced reactors
- Evaluation of candidate CTFD codes by applying those data.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/ROK

ABSTRACT

Development of Enhanced Reactor Operation Strategy through Improved Sensing and Control at Nuclear Power Plants

Primary Investigator (U.S.): David Holcomb, Oak Ridge National Laboratory (ORNL)

Primary Investigator (Republic of Korea): MG Na, Chosun University

Collaborators: Ohio State University (OSU); KAERI; Cheju University

Project Number: 2002-020-K

Project Start Date: December 11, 2001

Project End Date: December 30, 2004

The project proposes that the Oak Ridge National Laboratory (ORNL), The Ohio State University (OSU), the Korea Atomic Energy Research Institute (KAERI), the Chosun University (CU) and the Cheju National University (CNU) collaborate to examine, develop, and demonstrate how modern sensing and control can improve the operation of nuclear power plants.

A more precise knowledge of the reactor system state (e.g., primary coolant temperature, core flux map, primary and feedwater flowrates) can facilitate operation closer to design margins, improved thermal efficiency, and extended fuel burn-up. As a result, advanced control methods (e.g., innovative control algorithms) need to be developed to realize the benefits offered by improved sensing capability.

The project consists of three tasks. The objective of the first task is to evaluate the basis for current reactor operation strategies including assessment of the state-of-the art for primary system measurement, investigation of the effects of measurements limitations on operational performances of existing NPPs, and identification of potential operational/ safety improvements resulting from improved measurement and control. The objective of the second task is to develop three advanced sensors; a solid-state in-core flux monitor, a Johnson noise thermometer and a magnetic flowmeter. The objective of the third task is to take advantage of the benefits of improved sensors by devising advanced reactor operational strategies that optimize core performance and permit reduced operational margins.

Although the primary focus will be application to Generation IV reactors, the results will also have applicability to current operating plants.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/ROK

ABSTRACT

Condition Monitoring through Advanced Sensor and Computational Technology

Primary Investigator (U.S.): Vincent Luk, Sandia National Laboratories (SNL)

Primary Investigator (Republic of Korea): J-T Kim, Republic of Korea Atomic Energy Research Institute (KAERI)

Collaborators: Seoul National University, Pusan National University, Chungnam National University

Project Number: 2002-021K

Project Start Date: December 11, 2001

Project End Date: December 30, 2004

Deployment of advanced condition monitoring systems offers the prospect of improved performance, assessment, and operation, simplified design, enhanced safety, and reduced overall cost of advanced and next generation nuclear power plants (NPPs). For advanced and next generation NPP designs, the opportunity exists to develop and implement real-time and continuous monitoring systems by integrating advanced sensor and computational technology into design and operational concepts. This research project encompasses an international collaborative effort to develop and demonstrate advanced sensor and computational technology for continuous monitoring of the condition of components, structures, and systems in advanced and next-generation NPPs. This project includes the investigation and adaptation of several advanced sensor technologies from Korean and U.S. National Lab research communities for application to the NPP industry. Also, the project team will develop advanced sophisticated signal processing, noise reduction, and pattern recognition techniques and algorithms from other fields, such as satellites, robotics, and radar systems, as well as evaluate encryption and data authentication techniques for the wireless transmission of sensor data. Selected sensors and computational techniques will be demonstrated on prototypical components, structures, and systems.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/ROK

ABSTRACT

In-Vessel Retention Technology Development Use for Advanced PWR Designs in the USA and Republic of Korea

In-Vessel Retention Strategy for High-Power Reactors

Primary Investigators (U.S.): Joy Rempe, Idaho National Engineering Environmental Laboratory (INEEL) and Theo Theofanous, University of California, Santa Barbara (UCSB)

Primary Investigators (Republic of Korea): KY Suh, Seoul National University (SNU) and SJ Oh, Korean Electric Power Research Institute (KEPRI)

Collaborators: Westinghouse Electric

Project Number: 2002-022-K (I & II)

Project Start Date: January 15, 2002

Project End Date: December 30, 2004

In-vessel retention (IVR) is a key severe accident management strategy adopted by some operating nuclear power plants and proposed for some advanced light water reactors (ALWRs) and next-generation (Gen IV) reactors. If there were inadequate cooling during a reactor accident, a significant amount of core material could become molten and relocate to the lower head of the reactor vessel, as happened in the TMI-2 accident. If it is possible to ensure that the vessel head remains intact so that relocated core materials are retained within the vessel, the enhanced safety associated with these plants can reduce concern about containment failure and associated risk. For example, the Westinghouse AP600's enhanced safety resulted in NRC approving the design without requiring certain conventional features common to existing LWRs. However, it is not clear that currently proposed external reactor vessel cooling (ERVC) without additional enhancements could provide sufficient cooling for high-power reactors (up to 1500 MWe). Application of the design recommendations obtained from this project will enhance the performance of ERVC and core catchers to ensure that IVR will be available for reactors up to 1500 MWe (encompassing all reactor designs incorporating these features).

IVR is a keystone of the severe accident management for Westinghouse's AP600 (advanced passive light water reactor) design. A successful IVR would terminate a severe accident, passively, with the core in a stable, coolable configuration (within the lower head), thus avoiding the largely uncertain accident evolution with the molten debris on the containment floor. The passive plant design concept is being upgraded by Westinghouse to the AP1000, a 1000 MWe plant that is very similar to the AP600. The severe accident management approach is very similar also, including IVR as the keystone feature, and initial evaluations indicate that this would be feasible at the higher power as well

Task I – In-Vessel Strategy for High-Power Reactors

In this Sub-Project (Task), specific recommendations will be developed to improve the margin for IVR in high-power reactors (up to 1500 MWe). The systematic approach applied to develop these recommendations combine state-of-the-art analytical tools and key U.S. and Korean experimental facilities. Recommendations will focus on modifications

2001-022-K (I & II continued)

to enhance ex-vessel reactor cooling (improved data, vessel coatings to enhance heat removal, and an enhanced vessel/insulation configuration to facilitate steam venting) and modifications to enhance in-vessel debris coolability (enhanced in-vessel core catcher configuration, thickness, and material). Improved analytical tools and experimental data will be used to evaluate options that could increase the margin associated with these modifications. This increased margin has the potential to improve plant economics (owing to reduced regulatory requirements) and increase public acceptance (owing to reduced plant risk). Because the design information available for Gen IV designs is insufficient, these studies will initially focus on the Korean Advanced Power Reactor –1400 MWe (APR-1400) design. However, margins offered by each modification will be evaluated such that results can easily be applied to a wide range of existing reactors, advanced reactor designs, and Gen IV reactor designs.

This task is essential to make significant improvements to the safety and economics of existing, advanced, and next-generation nuclear reactors, world-wide.

Task II - Development of IVR for Advanced PWR Designs in the USA and Korea

The research in this task is intended to support the IVR aspect of the AP1000 design and to do so in a timely fashion for the certification submittal expected to be made by mid-2002 to the US Nuclear Regulatory Commission. IVR is now also being pursued in Korea (KEPRI being the leader of these efforts), for their evolutionary APR1400, as well as the Korean Standard Nuclear Power Plant (KSNPP), a 1,000 MWe PWR.

IVR is assured if the thermal loading from the melt is exceeded by the coolability limit (the critical heat flux—CHF) on the outside, everywhere on the lower head of the reactor pressure vessel. They both depend strongly on the angular position around the lower head, so the “everywhere” refers to all angular positions, from the very bottom (position 0º) to the very edge (position 90º). From experience with AP600, we know that this thermal-load-to-coolability margin is most comfortable in the lower portions, and that it reaches a minimum value at the position that corresponds to the top of the molten fuel pool (typically around 80º). Also, we know that this margin decreases as reactor power (decay heat) increases. It is quite comfortable at 1933 MWt (AP600), but at 3400 MWt (AP1000) it drops to under 10%, so on the basis of existing technology, it may be hard to justify an IVR -based severe accident management approach. The work proposed herein is aimed to remedy this situation, and present tangible evidence that this is indeed feasible.

This aim is to be principally pursued by means of a design improvement on the flow path between the lower head and the thermal insulation. Work with a preliminary such design implemented in the ULPU facility shows that we can expect CHF improvements by at least 25%, over current state-of-the-art values as defined by the ULPU results obtained in support of AP600. If this could be fully demonstrated, the 10% margin noted above would increase to almost 30%, a much more reasonable basis for the robust conclusion sought. As an auxiliary objective we propose to extend available natural convection data, again the state of the art being defined by ACOPO experiments conducted in support of AP600, to include the Rayleigh number range relevant to AP1000 (about an order higher than the AP600), and to refine our understanding (and quantification) of the heat flux peaking at the upper edge of the molten pool. Thus, the principal components of the effort will involve the large-scale experiments: ULPU at full scale, and ACOPO at 1/2 scale, but prototypical Rayleigh numbers. In addition, there will be fundamental components involving small experiments and numerical simulations.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/ROK

ABSTRACT

Passive Safety Optimization in Liquid Sodium-Cooled Reactors

Primary Investigator (U.S.): James E. Cahalan,
Argonne National Laboratory (ANL)

Primary Investigator (Republic of Korea):
Dohee Hahn, Republic of Korea Atomic Energy
Research Institute (KAERI)

Project Number: 2003-002-K

Project Start Date: January 2003

Project End Date: December 2005

A three-year collaboration is proposed between Argonne National Laboratory (ANL) and the Korea Atomic Energy Research Institute (KAERI) to identify and quantify the performance of innovative design features in metallic-fueled, sodium-cooled fast reactor designs. The objective of the work is to establish the reliability and safety margin enhancement provided by design innovations offering significant potential for construction, maintenance, and operating cost reductions. The targeted cost reductions and safety performance improvements are directly responsive to the goals for Generation IV nuclear energy systems.

The proposed work includes a combination of advanced model development, analysis of innovative design and safety features, and planning of key safety confirmation experiments. The model development task provides for required improvements in prediction of reactor thermal-hydraulic performance; reactor fuel, cladding, coolant, and structural material temperatures; and resulting reactivity feedback effects. The safety analysis task provides for evaluation of the effectiveness and safety performance impacts of specific design features, with emphasis on passive safety mechanisms that augment and replace costly engineered safety systems. The third task investigates the safety and operational performance implications of a Brayton power conversion cycle utilizing supercritical carbon dioxide, and incorporating an innovative heat exchanger design. This cycle offers the potential for a significant increase in operating efficiency, and the new heat exchanger design allows elimination of the intermediate heat transfer loop. Finally, the fourth task provides for planning of a series of experiments designed to verify the safety margins available in metallic-fueled, sodium-cooled reactors for in-vessel retention of core melt debris and mitigation of the consequences of extremely low probability severe accidents. Verification of material behavior provided by these experiments will provide a basis for containment design simplification, and corresponding cost reduction.

Each of the tasks proposed here is thus specifically aimed at identifying and accurately quantifying the safety and operational performance benefits of innovative design features in metallic-fueled, liquid-sodium-cooled fast reactor, with the objective of simplifying reactor and plant designs and reducing costs.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/ROK

ABSTRACT

Developing and Evaluating Candidate Materials for Generation IV Supercritical Water

Primary Investigator (U.S.): James I. Cole
Argonne National Laboratory (ANL)

Primary Investigator (Republic of Korea):
J. Jang, Republic of Korea Atomic Energy Research Institute (KAERI), and S.H. Hong, Republic of Korea Advanced Institute of Science and Technology (KAIST)

Collaborators: Republic of Korea Advanced Institute of Science and Technology, University of Michigan, Idaho National Engineering and Environmental Laboratory, University of Wisconsin

Project Number: 2003-008-K

Project Start Date: January 2003

Project End Date: December 2005

The goal of this project is to establish candidate materials for supercritical water reactor (SCWR) designs and to initiate the evaluation of the mechanical properties, dimensional stability, and corrosion resistance. To overcome the principal technical and scientific obstacles to the long-term future use of nuclear energy, new reactor designs must offer enhanced safety and reliability, sustainability and economics. To meet these goals, Generation IV (GEN IV) reactor designs must incorporate advanced materials for cladding and structural components. Currently, insufficient physical property data exist to qualify candidate materials. In many cases, candidate materials have not even been identified. For all GEN IV designs, significant materials property data must be obtained to license future reactor designs.

To meet the goals of the GEN IV Reactor research initiative, international collaborations are critical in terms of shared resources and shared expertise. Because of the significant costs associated with nuclear systems research an international cost sharing approach will provide maximum value for the limited research dollars. Both the Republic of Korea (ROK) and the United States (US) have a shared interest in the development of advanced reactor systems that employ supercritical water as a coolant.

SCWRs are one of the more promising GEN IV nuclear systems concepts due to enhanced thermal efficiencies and relative compactness when compared to current light water reactor (LWR) technology. The relatively mature alloy development programs for supercritical fossil plants (SC-FP) can be used as a baseline for the development of fuel cladding and structural materials in a SCWR. The SC-FP alloys have known corrosion resistance properties but have not been evaluated relative to degradation in radiation fields. Additionally, materials developed for the fast reactor programs, which operated in similar temperature regimes as SCWR, will also be evaluated for SCWR applications. These alloys have known radiation resistance, but the corrosion performance is unknown. To understand the relative materials compatibility, a comprehensive research program is proposed that initially evaluates state-of-the-art SC-FP and fast reactor materials for application in SCWR, and expands on these alloys to produce materials optimized for SCWR fuel cladding and core internal structures.

2003-008-K (continued)

An integrated research program involving Argonne National Laboratory (ANL), The University of Michigan (UM), The Idaho National Engineering and Environmental Laboratory (INEEL), the University of Wisconsin (UW), Korea Atomic Energy Research Institute (KAERI), and Korea Advanced Institute of Science and Technology (KAIST) is proposed to advance goals of the SCWR concept and to aid in the removal of technical barriers regarding fuel cladding and reactor core internals performance. The program will focus on the selection and qualification of advanced materials for SCWR applications. Three materials classes will be investigated, 1) ferritic and ferritic-martensitic steels, 2) austenitic alloys 3) and developmental alloys such as oxide dispersion strengthened (ODS) alloys, nanocrystalline alloys, and grain boundary engineered alloys.

The research program will comprise three phases, with the findings of each phase constituting a deliverable which can be readily disseminated to organizations involved in the design and development of the SCWR concept as well as other reactor concepts where advanced high temperature alloys are required. Phase I of the project will involve a comprehensive literature search to identify the most promising candidate alloys for further investigation. In Phase II of the project, each of candidate alloys will be evaluated in terms of high temperature creep strength, stress corrosion cracking susceptibility, radiation stability and weldability. Finally, the third phase of the project will provide material recommendations and an overall reactor irradiation plan for future GEN IV programs. Although this project will not be able to perform a complete qualification of materials for SCWR application, it will generate data to support a recommendation of prime candidate materials for further evaluation and inreactor testing.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/ROK

ABSTRACT

Development of Safety Analysis Codes and Experimental Validation for a Very High Temperature Gas-Cooled Reactor

Primary Investigator (U.S.): Dr. Chang H. Oh,
Idaho National Engineering and Environmental Laboratory

Primary Investigator (Republic of Korea):
Professor Hee Cheon No, Republic of Korea
Advanced Institute of Science and Technology (KAIST)

Collaborators: University of Michigan

Project Number: 2003-013-K

Project Start Date: January 2003

Project End Date: December 2005

The proposed research focuses on development of new **Advanced Computational Methods** for safety analysis codes for Very High Temperature Gas-Cooled reactors (VHTGR), and **numerical & experimental validation of these computer codes.** The research proposes to improve two wellrespected light water reactor transient response codes (RELAP5/ATHENA and MELCOR) in the modeling of molecular diffusion and chemical equilibrium, and to validate these codes against VHTGR accident data, i.e., air ingress and others from literature. The VHTGR is intrinsically safe, has proliferation resistant fuel cycle, and many advantages relative to light water reactors (LWRs). This study consists of six tasks: (a) development of computational methods for VHTGR, (b) theoretical modification of aforementioned computer codes for molecular diffusion (RELAP5/ATHENA) and modeling CO and CO_2 equilibrium (MELCOR), (c) development of state-of-the-art methodology for VHTGR neutronic analysis and calculation of accurate power distributions and decay heat deposition rates, (d) reactor cavity cooling system experiment, (e) graphite oxidation experiment, and (f) validation of these codes.

The VHTGRs are those concepts that have average coolant temperatures above 900^{o}C or operational fuel temperatures above 1250^{o}C. These concepts provide the potential for increased energy conversion efficiency and for high-temperature process heat application in addition to power generation. While all the High Temperature Gas Cooled Reactor (HTGR) concepts have sufficiently high temperature to support process heat applications, such as coal gasification, thermochemical **hydrogen production**, desalination or cogenerative processes, the VHTGR's higher temperatures allow broader applications. However, due to the high temperature operation, this reactor concept can be detrimental if accidents occur by a loss-of-coolant accident (LOCA) or a pipe breaks due to seismic activities and others. Following the loss of coolant through the break and coolant depressurization, air will enter the core through the break by molecular diffusion and ultimately by natural convection, leading to oxidation of the in-core graphite structure and fuel. The oxidation will accelerate heatup of the reactor core and the release of toxic gasses (CO and CO_2) and fission products. Thus, without any effective countermeasures, a pipe break may lead to significant fuel damage and fission product release. As of today, the world does not have reliable numerical tools to analyze this event. The INEEL has investigated this event for the past three years for the HTGR. The new code development, improvement of these codes, and experimental validation are imperative to narrow the gap between predicted knowledges on this type of accident and the real phenomena occurring in the reactor.

2003-013-K (continued)

This project promotes the development of advanced numerical schemes and technologies that will enhance the safety and economics of a range of reactor designs. Innovative concepts, methodology, and data that will be obtained from this study include:

- development of a benchmark safety code.
- incorporation of diffusion model into RELAP5/ATHENA code.
- incorporation of chemical equilibrium model into MELCOR code
- development of state-of-the-art methodology for VHTGR neutronic analysis and calculation of accurate power distributions and decay heat deposition rates
- code validation using data to be collected in this study and additional data from German NACOK or Chinese HTR-10 experiments.

This project will validate computational methods using new experimental data to be collected in 2003 and data from Germany or China. The most significant issue for the U.S. Nuclear Regulatory Commission (NRC) licensing of VHTGRs is the V&V of computer codes used in the neutronic and safety analysis of plant performance. At present, such capability is extremely limited. Validation of the wellknown computer codes will facilitate the licensing process.

Project tasks have been defined to take advantage of key capabilities of this international team. Our highly esteemed and experienced experts in Korea on the high temperature gas cooled reactor system bring code development, scaling test and relevant tests for this project that enable production of quality work. In support of DOE programs and of the nuclear power industry, the INEEL has long been an international leader in treating transient reactor thermal hydraulic behavior, both experimentally and numerically. Based on its large-scale experiments at the Water Reactor Research Test Facility, INEEL has developed the world's leading code (RELAP5/ATHENA) for transient analyses of hypothesized reactor accident scenarios. That same experimental expertise is employed for this project. In addition, the INEEL's long history of collaboration with international and academic research organizations ensures a strong research team as a leading organization.

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/ROK

ABSTRACT

Advanced Corrosion-Resistant Zr Alloys for High Burnup and Generation IV Applications

Primary Investigator (U.S.): Arthur T. Motta, Pennsylvania State University

Primary Investigator (Republic of Korea): Yong Hwan Jeong, Republic of Korea Atomic Energy Research Institute (KAERI)

Collaborators: Westinghouse Electric Company, LLC, University of Michigan, Hanyang University

Project Number: 2003-020-K

Project Start Date: January 2003

Project End Date: December 2005

A collaborative United States/Korea research program to develop Zr alloys for advanced nuclear fuel designs is proposed. In addition to Pennsylvania State University, the U.S. collaborators include Westinghouse Electric Company and the University of Michigan. The Korean collaborators are the Korea Atomic Energy Research Institute (KAERI) and Hanyang University. The objective of the program is to develop advanced, corrosion-resistant Zr alloys for extreme environments, focusing specifically on (i) high burnup applications in current light water reactors (LWRs) and (ii) cladding and reactor internal components in the supercritical water reactor (SCWR), a Generation IV reactor concept. Developing such alloys will permit higher duty operation of current fuel as well as fuel for new reactor designs targeted for near term deployment. In addition, development of corrosion-resistant Zr alloys will provide greater design flexibility and allow for economies of operation in the SCWR.

The proposed program builds on two highly successful NERI programs, which are now ending, and includes several world-class experts in Zr alloy corrosion and irradiation effects from both Korea and the United States. The collaborating organizations include, a major fuel vendor, three major research universities (two in the United States and one in Korea), and an internationally recognized research organization (KAERI). The proposed program will also employ the resources and expertise available at a major user facility at a U.S. national laboratory (the Advanced Photon Source at Argonne).

Waterside corrosion and the associated hydrogen pickup can be a limiting factor in the operation of Zr-based fuel cladding in current light water reactors (LWRs) and will be an important concern in future evolutionary and revolutionary designs called for under the Generation IV Reactor Initiative. In order to meet the more stringent economic demands of nuclear technology, advanced LWRs, and Generation IV reactor concepts, materials must operate under more severe conditions. Fuel cladding and structural materials must be able to perform at higher fluences, higher operating temperatures, longer residence times, and higher burnups than current operating limits. Therefore, it is crucial to improve the corrosion resistance of zirconium alloys for both near-term and long-term applications.

The first step in improving corrosion resistance is developing a clear understanding of the mechanisms of corrosion. During a previous NERI program, a combination of detailed characterization studies and modeling was used to identify some of the crucial parameters that govern corrosion behavior. These detailed studies, which are summarized in the proposal, were performed on complex commercial alloys. The studies provided valuable insight to the corrosion process but

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identification of individual mechanisms was difficult. Accurate determination of mechanisms must come from model binary and ternary alloys that are specifically designed to isolate the effects of individual parameters on the corrosion process.

The objective of this program is to develop and demonstrate a technical basis for improving the corrosion resistance of zirconium-based alloys in aqueous reactor coolants. In particular, the goal of the proposed research is to develop Zr-based alloys with superior corrosion resistance relative to the current state-of-the-art alloys used in LWRs, namely, Zircaloy-2, Zircaloy-4, ZIRLO, and the Zr-Nb alloys containing 1.0 and 2.5% Nb. These existing commercial alloys were formulated largely through empirical methods of alloy addition, testing, evolutionary optimization of composition and thermo-mechanical processing. Incremental improvements using this classical approach are probably still possible. However, a more fundamental understanding of the effect of alloy chemistry and microstructure on the structure and degradation of the protective barrier oxide is necessary to achieve significant improvements in corrosion resistance. The focus of the proposed approach, therefore, is to characterize the effect of individual chemical and metallurgical variables in selected alloys on oxide properties and to identify those factors that significantly reduce the corrosion rate. This knowledge will serve as the basis for the design of new alloy compositions and processing routes.

Specifically, a series of *model alloys* will be prepared by vacuum arc melting small button ingots that will be reduced to strip by thermo-mechanical processing and autoclave tested. Two series of model alloys will be manufactured and tested. The first series is designed to elucidate the role of solute atoms in the Zr matrix on the corrosion rate (focusing on valence effects and solute concentration). The second series is designed to elucidate the role of precipitates in the corrosion process (focusing on precipitate size, volume fraction, and precipitate type).

These alloys will be tested in different autoclave environments to determine the growth kinetics of the protective oxide and the oxide thickness at transition. These oxides will be characterized using an array of advanced characterization techniques to determine the relationship between oxide microstructure and the two parameters that control corrosion rates (oxygen transport and transition thickness). These characterization techniques include synchrotron radiation microbeam x-ray diffraction and fluorescence (techniques developed at the Advanced Photon Source at Argonne in a current NERI program), cross sectional transmission electron microscopy (TEM), oxide stress measurements, and nano-indentation. The synchrotron techniques provide unique and hitherto unobtainable information about the oxide that, combined with the detailed TEM and mechanical characterization, will provide a much more complete and detailed picture of the oxide microstructure than has been obtained to date. Such data will allow the detailed mechanistic modeling of corrosion.

The analysis of the experimental results will help answer fundamental questions related to the corrosion process. In particular, the program will develop a scientific basis for the well-known empirical correlations between corrosion rates and alloy chemistry/processing and investigate the operating range of Zr alloys at high temperature. The scientific and technical benefit of this program will come from an increased ability to predict corrosion behavior and in the availability of alloys that exhibit superior corrosion performance under severe duty cycles in current and advanced LWRs and in the SCWR.

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INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/ROK

ABSTRACT

Development of Structural Materials to Enable the Electrochemical Reduction of Spent Oxide Nuclear Fuel in a Molten Salt Electrolyte

Primary Investigator (U.S.): James J. Laidler,
Argonne National Laboratory (ANL)

Primary Investigator (Republic of Korea):
Seong Won Park, Republic of Korea
Atomic Energy Research Institute (KAERI)

Project Number: 2003-024-K

Project Start Date: January 2003

Project End Date: December 2005

This ANL- and KAERI-led collaborative project is designed to develop advanced structural materials to enable the electrolytic reduction of spent oxide nuclear fuel in a molten salt electrolyte. This will include the selection and testing of commercial alloys and ceramics as well as the engineering and testing of customized materials systems.

The electrolytic reduction of spent oxide fuel involves the liberation of oxygen in a molten LiCl electrolyte, which results in a chemically aggressive environment that is too corrosive for typical structural materials. Even so, the electrochemical process vessel, structural cell components, and the electrical supply materials must each be resilient in the presence of oxygen, the molten salt components, and various impurities at 650ºC to enable high processing rates and an extended service life.

The goals of this program are to:

1. Assess and select candidate materials for service in the electrolytic reduction process vessel.

2. Develop new candidate material systems (e.g., functional barrier coatings) for service in the electrochemical reduction process vessel.

This project provides a necessary component to the implementation of electrolytic reduction technology, but it does not deal directly with the mechanisms or operations of the process. The materials solutions developed here will benefit the “Advanced Fuel Cycle Initiative (AFCI) program” of the United States Department of Energy for the reduction of transuranic and fission product oxides and the “Advanced Spent Fuel Conditioning Process (ACP)” of the Korea Atomic Energy Research Institute for the conditioning of spent fuel for long-term storage and eventual disposal. The successful implementation of this project will provide an enabling solution for the effective management of spent fuel, and contribute to the establishment of a nuclear fuel cycle technology that is proliferation resistant and cost effective.

This project is proposed as collaboration between Argonne National Laboratory (ANL) and the Korea Atomic Energy Research Institute (KAERI). The primary contributing organizations within the two parties will be the Chemical Technology Division at ANL and the Advanced Spent Fuel Conditioning Process R&D group at KAERI.

Other Collaborations

U.S. DEPARTMENT OF ENERGY
INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE
DOE/Other Collaborations

ABSTRACT

Melt Coolability and Concrete Interaction (MCCI)

Primary Investigator (U.S.): M. T. Farmer and J. L. Binder, Argonne National Laboratory (ANL)

International Organization: Organization for Economic Cooperation and Development (OECD)/ Nuclear Energy Agency (NEA)

Project Number: 2002-001-N

Project Start Date: March 1, 2002

Project End Date: December 30, 2005

The MCCI program is a collaborative project between the Reactor Analysis and Engineering (RAE) Division of Argonne National Laboratory, the U.S. Nuclear Regulatory Commission, and a consortium of ten international participants represented by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development. U.S. NRC is the Operating Agent for the program.

Although extensive research has been conducted over the last several years in the areas of melt coolability and core-concrete interaction, two important issues warrant further investigation. The first issue concerns the effectiveness of water in terminating a core-concrete interaction by flooding the interacting masses from above, thereby affecting a quench of the molten core debris and rendering it permanently coolable. The second issue concerns long-term two-dimensional ablation by a prototypic core oxide melt. The goal of the MCCI research program is to conduct reactor material experiments and associated analysis to achieve the following two technical objectives:

1. Resolution of the ex-vessel debris coolability issue through a program which focuses on providing both confirmatory evidence and test data for the coolability mechanisms identified in MACE integral effects tests.

2. Address remaining uncertainties related to long-term two-dimensional core-concrete interaction under both wet and dry cavity conditions.

Achievement of these two main objectives will lead to improved accident management guidelines for existing plants and also better containment designs for future plants.